Calculation of Neutronic Parameters in Support of a BOR-60 Experimental FA with Moderating Elements

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Abstract. At present, different nuclear fuels (NF) to be used in advanced fast neutron reactors (AFR) are tested in the BOR-60 reactor. In such in-pile testing the top priority is to ensure the maximum possible compliance of the target NF irradiation parameters with the design operating parameters. The key monitored parameters in testing experimental fuel elements are the fuel burnup rate and linear heat rate that depend on the nuclear fission rate in the fuel elements. Rather low enrichment of tested fuel compositions (as compared to BOR-60 standard fuel) and low neutron flux density (as compared to big fast neutron reactors) in the BOR-60 core make it difficult or even impossible to provide the target heat and fuel burnup rates of NF. To increase the nuclear fission reaction rate in the experimental fuel elements it is suggested to install neutron moderating elements in an experimental fuel assembly (EFA). The calculated data analysis confirmed the effectiveness and safety of the suggested design solution. This EFA design option enables wider BOR-60 capabilities in testing advanced nuclear fuels due to high fuel burnup and heat rates.

Key words: BOR-60 reactor, fuel element, heat rate, moderator.

1. Introduction

Nowadays, different types of nuclear fuel (NF) intended for advanced fast reactors are being tested in the BOR-60 reactor. When performing the in-pile tests, the primary concern is to provide the required test conditions appropriate to the maximum to the operation conditions for a specific type of nuclear fuel. The most important parameters to test experimental fuel elements are the NF burnup rate and linear heat rate; these parameters depend on the fuel nuclei fission in the fuel elements.

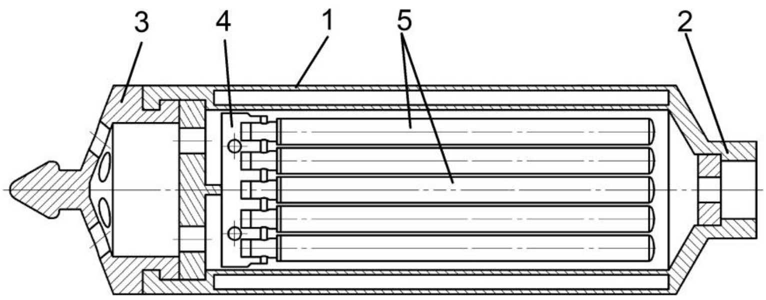
The enrichment of fuel compositions under testing in fissile nuclides (10÷20%) is, as a rule, lower than the enrichment of the standard BOR-60 fuel (~70%). The BOR-60 neutron flux density is, on the contrary, 1.5-2 times lower as compared to the power and demonstration reactors under design, of which fuels are to be tested. As a result, the nuclei fission rate in the experimental fuel elements is relatively low and does not provide for the required heat rate and NF burnup rate.

However, the fission rate can be raised by decreasing the neutron energy. It can be done by inserting a neutron moderator close to the fuel elements in question.

The purpose of work is to conduct calculations in justification of the efficiency and safety of a new EFA design with a neutron moderator.

2. EFA design

To irradiate different nuclear fuels in the BOR-60 reactor, a dismountable EFA is used, of which design is described in [1] and presented in Fig. 1.



1 – duct, 2 – fixture, 3 – dismountable head,
4 – spacer grid, 5 – bundle of fuel elements
FIG. 1. EFA cross-section.

To increase the nuclei fission rate in the experimental fuel elements, it is proposed to replace a part of fuel elements in the EFA with a neutron moderator. Zirconium hydride is considered to be the moderator material since it has a high moderating capability and is successfully used in the BOR-60 reactor for many years to locally soften the neutron spectrum [2, 3].

3. Neutronic calculations

In justification of the EFA design with a moderator, MCU-RR-based calculations were done [4]. In doing so, an automated calculation complex KAR [5] was used to create a 3D BOR-60 model corresponding to a current reactor core arrangement given in Fig.2. The EFA was modeled in cell E17, 5th row.

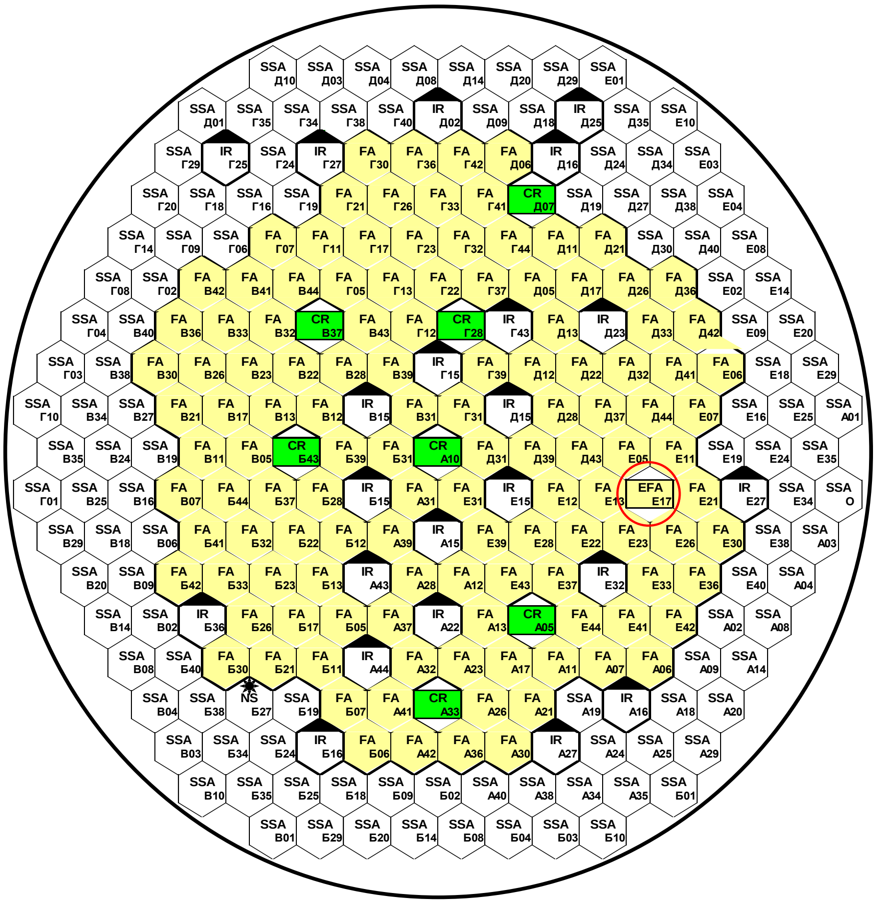
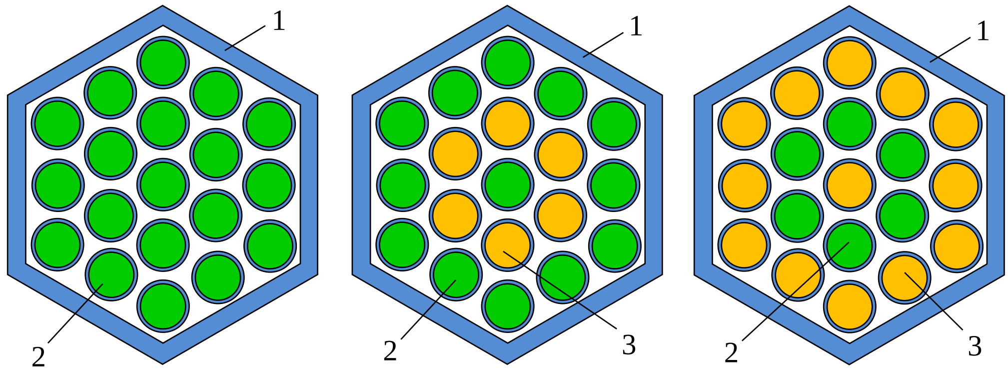


FIG. 2. BOR-60 core arrangement.

Three EFA design options were considered (Fig. 3):

- Option 1: a 19-element EFA with no moderator (Fig. 3a);
- Option 2: a 13-element EFA with 6 moderating elements (Fig. 3b);
- Option 3: a 6-element EFA with 13 moderating elements (Fig. 3c).



1 – duct, 2 – fuel element, 3 – moderating element

Fig. 3. EFA cross-section.

The modeling was done for the fuel elements (claddings 6.9mm in diameter and 04mm thick) with oxide uranium fuel 20% enriched in ^{235}U and 9.0g/cm^3 density. The moderating elements are of the same design as fuel ones but with a moderator instead of fuel. The moderator material is granulated vibropacked zirconium hydride ($ZrH_{1.85}$) with an effective density of 4.1g/cm^3.

Figure 4 presents the neutron spectra (26 groups BNAB) [6] averaged over the fuel part of the EFA fuel elements. Presented spectrum are normalized to unity.

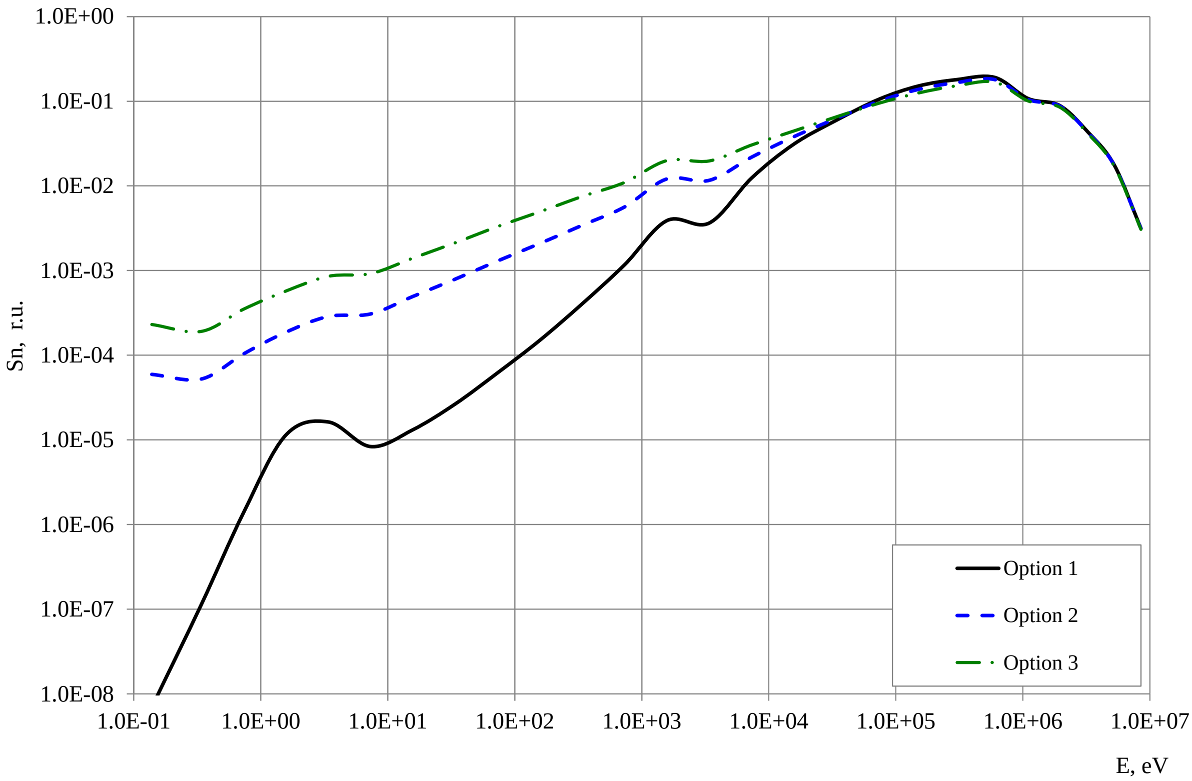


FIG. 4. Neutron spectra (26 groups BNAB) in the fuel elements.

The calculation results show the moderating elements allow increasing significantly the amount of thermal and epithermal neutrons at the experimental fuel elements location. At that, the amount of fast neutrons (with E > 0.1MeV) in the flux density decreases insignificantly. So, for the first option, the amount of fast neutrons within the fuel part ranges within 75-80%, while for options 2 and 3 it ranges within 70-75% and 65-70%, respectively.

Figure 5 presents axial distribution of linear power (Q_l) for all three EFA options normalized for the reactor thermal power of 50 MW. Since the fuel enrichment in the EFA fuel elements is significantly lower than the BOR-60 standard one, then to calculate the heat rate, a technique was used accounting the contribution of delayed gamma-emission from fission products [7, 8].

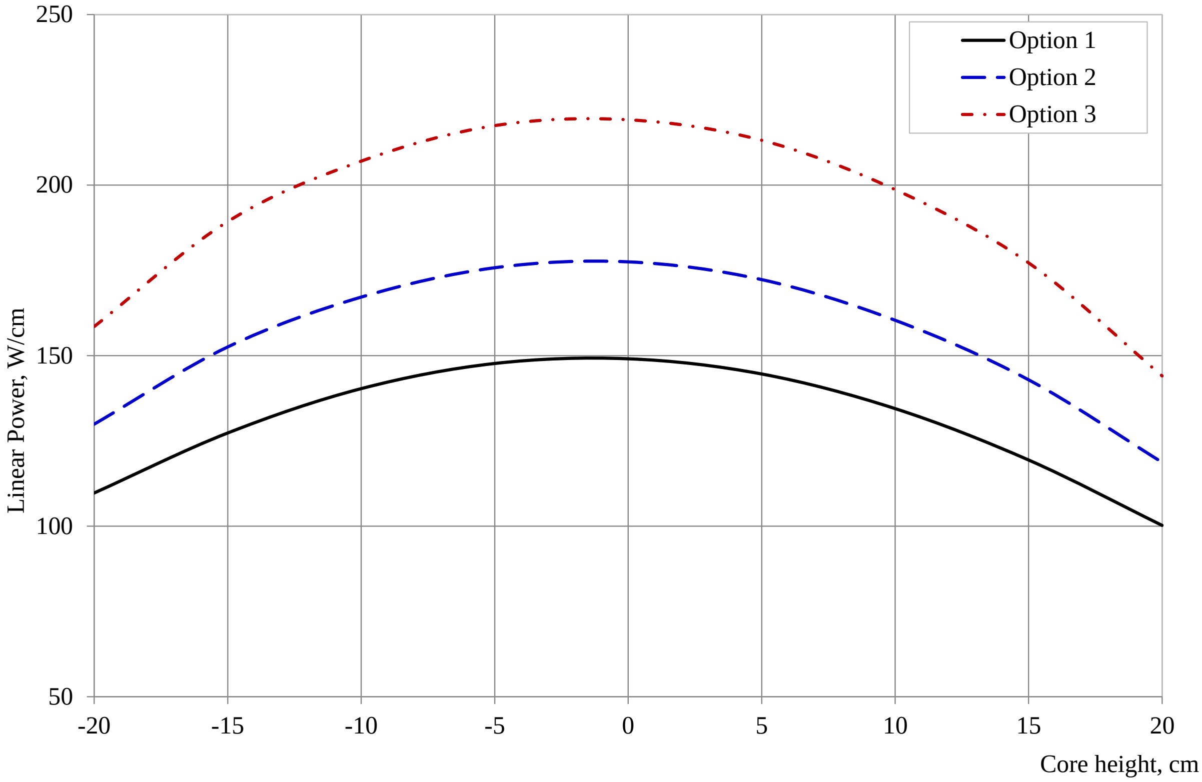


FIG. 5. Axial distribution of linear heat rate.

It is seen from the above Figure that the moderating elements in the EFA allow a significant heat rate increase in the experimental fuel elements. So, the linear heat rate for options 2 and 3 is about 20% higher as compared to option 1. Thus, the fuel burn up rate increases as well in the fuel elements that allows for accelerated tests.

Table 1 gives the results of the neutronic calculations for three EFA options: amount of fast neutrons (E > 0.1 MeV) in the total flux density ($F_{0.1}$), damage dose accumulation rate in steel (D), average neutron energy (E_{av}), maximal linear power of fuel elements (Q_l^{max}) and reactivity (ρ) relative the EFA with no moderator. The calculations were one for the core mid-plane.

TABLE I: NEUTRONIC CALCULATION RESULTS.

Option	$F_{0.1}$, %	D, dpa/s	E_{av}, keV	Q_l^{max}, W/cm	ρ, %Δk/k
1	78	9.17E-7	289	150	0.0
2	74	9.04E-7	150	180	0.1
3	69	8.85E-7	79	220	0.2

To investigate the effect of moderator on the heat rate non-uniformity over the EFA fuel elements cross-sections, neutronic calculations were done. For this purpose, an EFA was modeled to have 7 fuel and 12 moderating elements (Fig.6). The heat rate non-uniformity was investigated for three fuel elements (marked 1 through 3 in the Figure):

- Fuel element #1 – the closest to the core center;
- Fuel element #2 – central;
- Fuel element #3 – the farthest from the core center.

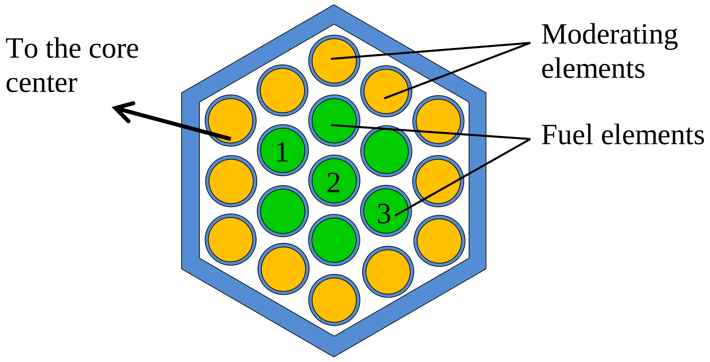


FIG. 6. EFA cross-section to calculate the heat rate non-uniformity

In the calculation model, the fuel in each o f three fuel elements in question was broke up into 8 layers over the radius and into 12 sectors in azimuth (Fig.7). The radial layers were of different thickness becoming less with distance from the fuel element center. This is to fix a probable effect of a local increase in heat rate in the thin layer at the fuel column boundary that is typical for thermal reactors.

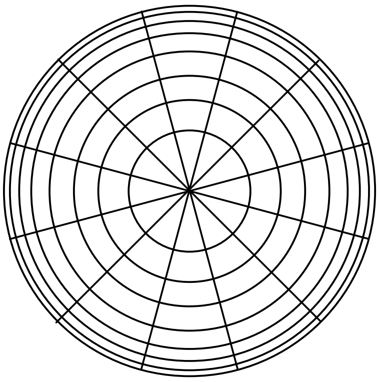


FIG. 7. Computational mesh for the heat rate over the fuel element cross-section.

Figure 8 shows the calculation results of the relative specific heat rate over the fuel elements cross-sections. The plots are given that show the maximal heat rate non-uniformity for each fuel element in question. The average specific heat rate over the fuel element cross-section was taken as a unit.

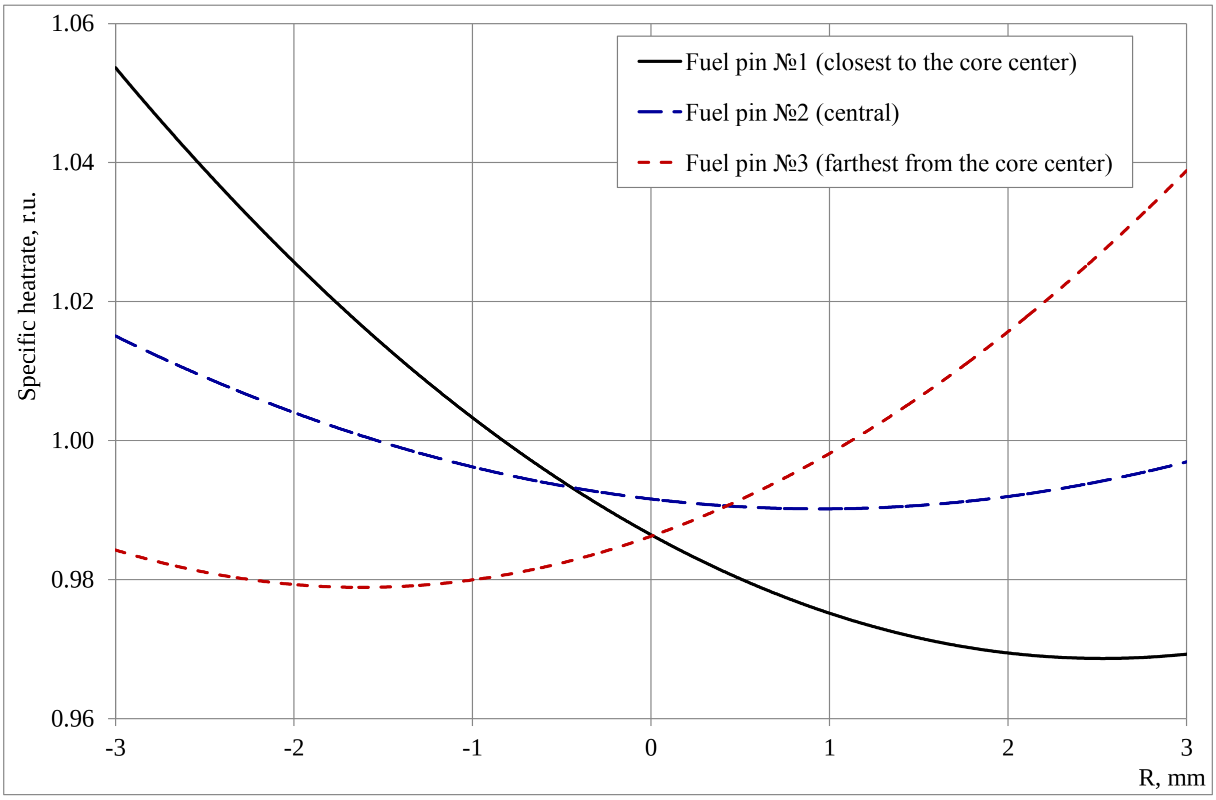


FIG. 8. Distribution of the specific heat rate over the fuel elements cross-sections.

Some increase in the heat rate was observed from the moderator; no significant heat rate splashes were observed on the fuel column surface. The non-uniformity factor (max value relative to the average one) of the specific heat rate distribution over the fuel element cross-section did not exceed 1.06.

The efficient moderator in the EFA increases the fission rate not only in the EFA fuel element but also in standard FA fuel pins located around. The effect of the EFA with a moderator on the heat rate in the standard FAs was estimated. The EFA with a moderator was modeled in BOR-60 cell A22, 4th row (Fig.2). A fresh standard FA was modeled in the neighboring cell A12 located closer to the core center. The heat rate was calculated at the core mid-plane level. The distribution of relative heat rate over the fuel elements is shown in Figure 9. The maximal allowable linear power for the BOR-60 standard fuel pins was taken as a unit.

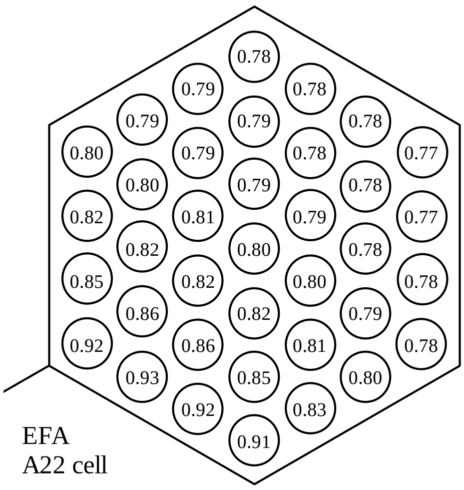


FIG. 9. Distribution of relative heat rate over fuel elements.

The calculation results show the limited linear heat rate was not exceeded even for the fuel pins located close to the EFA with a moderator.

So as to verify the calculation results by the experimental data, the EFA with a moderator was inserted into the instrumented BOR-60 cell D-23, 5^{th} row, to measure the coolant heating. Thermocouples were installed above and under the fuel elements to record the coolant temperature, its flow rate through the EFA being known.

The heating was calculated using software ANSYS CFX [9]. The EFA geometry and composition of its material were modeled according to the design documents. The thermo-physical properties of the material used in the model were tested in the similar calculations done for different BOR-60 irradiation rigs and FAs.

The calculated and experimental value of coolant heating (at a nominal reactor power) coincided within the uncertainty and made up 196°C and 201°C, respectively, that confirms a high precision of the heat rate calculation results for the EFA with a moderator.

4. Conclusion

The application of the proposed EFA design enlarges significantly the BOR-60 capabilities in testing promising nuclear fuels.

The described technical solution [10] allows a wide range of required irradiation conditions to be provided for the experimental fuel elements, including:

- nuclear fuel burnup rate;
- linear power;
- neutron spectrum;
- damage dose to burnup ratio.

If necessary, the neutron spectrum can be softened significantly due to:

- more moderating elements in the EFA;
- increase of the moderating element diameter (as much as allowed by the fuel element geometry and EFA inner space);
- zirconium hydride as a moderator in the form of pellets/rodlets of 5.4 g/cm^3 in density.

At present, the proposed EFA design is used to test promising fuels in the BOR-60 reactor.

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